Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

March 20, 2007

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

10 CFR 50.73

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - LICENSEE EVENT REPORT (LER) 50-328/2007-001-00

The enclosed LER provides details concerning an automatic reactor trip and engineered safety feature (ESF) actuation of auxiliary feedwater. The automatic trip occurred as a result of low-low steam generator level when a feedwater regulating valve closed as a result of a failed control air line. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv) as an event that resulted in the automatic actuation of engineered safety features, including the reactor protection system.

Sincerely,

Glenn W. Morris
Manager, Site Licensing and
Industry Affairs

Enclosure
cc (Enclosure):
INPO Records Center
Institute of Nuclear Power Operations
700 Galleria Parkway, SE, Suite 100
Atlanta, Georgia 30339-5957

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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Sequoyah Nuclear Plant (SQN) Unit 2	05000328	1 OF 5

4. TITLE
Reactor Trip Following Closure of Main Feedwater Valve Due To Control Air Line Failure

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
01	23	2007	2007	- 001 -	00	03	20	2007	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
100	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
N. R. Thomas, Nuclear Engineer	423-843-7749

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On January 23, 2007, at 1244 Eastern standard time (EST) with Unit 2 operating at 100 percent power, an automatic reactor trip occurred because of a low-low steam generator level on Loop 2. The immediate cause was closure of the Loop 2 main feedwater regulating valve as a result of a failed control air line. At approximately 1030 EST, it was identified that the air line tubing to the Loop 2 main feed regulating valve appeared to be damaged and leaking. Operations discussed the condition with plant management and agreed to pursue dogging the valve open so maintenance could be performed online. During the evolution to open and place the bypass regulating valve in automatic, a swing in control signal was seen and feedwater flow decreased rapidly. The SRO directed that a manual reactor trip be initiated. The automatic trip occurred before the manual trip was initiated. Following the reactor trip, the plant systems responded as designed. The unit entered Mode 3 and an event investigation was initiated. The feedwater regulating valve's control air line was damaged as a result of improper routing of field tubing during a recent outage modification. The routing of the control air line did not sufficiently account for movement of the valve due to thermal growth.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

I. PLANT CONDITION(S)

Unit 2 was operating at 100 percent power when the reactor trip occurred.

II. DESCRIPTION OF EVENT

A. Event:

On January 23, 2007, at 1244 Eastern standard time (EST) with Unit 2 operating at 100 percent power, the reactor tripped as a result of low-low steam generator (EIIS code AB) level on Loop 2. The immediate cause was closure of the Loop 2 main feedwater regulating valve (EIIS code SJ) because of a failed control air line. The feedwater regulating valve's control air line was damaged resulting from improper routing of field tubing during a recent outage modification. During an attempt to place the bypass feedwater regulating valve in control in order to allow repair to the damaged control air line, the control air line to the main feedwater regulating valve broke. The main feedwater regulating valve failed closed as designed upon loss of control air.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

January 23, 2007 at 10:30 EST	Operations was notified by Maintenance personnel of a significant air leak at the Loop 2 feedwater regulating valve.
January 23, 2007 at 11:30 EST	Operations conducted a briefing for "dogging" Loop 2 feedwater regulating valve in accordance with system operating instructions to allow repair of the damaged control air line.
January 23, 2007 at ~12:35 EST	Operations placed the Loop 2 bypass regulating valve level controller in manual and began opening.
January 23, 2007 at 12:43 EST	Entered Limiting Condition for Operation 3.7.1.6 while attempting to dog valve Loop 2 feedwater regulating valve.
January 23, 2007 at ~12:43 EST	Operations placed the Loop 2 bypass regulating valve level controller in automatic with bypass valve open.
January 23, 2007 at 12:43 EST	Steam generator Loop 2 low level alarm was received.
January 23, 2007 at 12:44 EST	The SRO directed that a manual reactor trip be initiated. Steam generator Loop 2 low-low level automatic reactor trip occurred.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

D. Other Systems or Secondary Functions Affected:

No other systems or secondary functions were affected by this event.

E. Method of Discovery:

Prior to the reactor trip, a Maintenance engineer performing a walk down noticed a control air leak on the Loop 2 feedwater regulating valve. Operations immediately began preparations to dog the valve so that the air leak could be repaired.

During the evolution to open and place the bypass regulating valve in automatic, a swing in control signal was seen and feedwater flow decreased rapidly.

F. Operator Actions:

After the SRO directed that a manual reactor trip be initiated, an automatic trip occurred. Control Room personnel responded as prescribed by emergency procedures. They promptly diagnosed the plant condition and took actions necessary to stabilize the unit in a safe condition and maintained the unit in hot standby, Mode 3.

G. Safety System Responses:

The plant responded to the reactor trip as designed.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of the event was closure of the Loop 2 main feedwater regulating valve as a result of a failed control air line. The closure of the feedwater regulating valve resulted in a reactor trip from low steam generator level.

B. Root Cause:

The feedwater regulating valve's control air line was damaged as a result of improper routing of field tubing during a recent outage modification. The routing of the control air line did not sufficiently account for movement of the valve due to thermal growth. During an attempt to place the bypass feedwater regulating valve in control in order to allow repair to the damaged control air line, the control air line to the main feedwater regulating valve broke. The main feedwater regulating valve failed closed as designed upon loss of control air.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

C. Contributing Factor:

There were no contributing factors.

IV. ANALYSIS OF THE EVENT

The plant systems responded to the reactor trip as designed. The reactor coolant system (RCS) average temperature was near 578.2 degrees F prior to the loss of main feedwater. When feedwater was lost, RCS average temperature made a slight increase before the reactor trip. Following the reactor trip, the loss of nuclear heat generation resulted in a rapid decrease in RCS average temperature to 535 degrees F. As heat removal in the steam generators decreased as a result of increased steam pressure, the decrease in RCS temperature slowed. The introduction of cold auxiliary feedwater (AFW) resulted in a slower, but continued reduction in RCS temperature until AFW flow was reduced about 10 minutes after the reactor trip. RCS temperature then started to increase. RCS temperature remained within Technical Specification limits and bounded by the Safety Analysis Report (SAR) analysis.

The plant responded as expected for the conditions of the trip. No Technical Specification limits were exceeded and the SAR analysis of this event remained bounding.

V. ASSESSMENT OF SAFETY CONSEQUENCES

Based on the above "Analysis of The Event," this event did not adversely affect the health and safety of plant personnel or the general public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

Control Room personnel responded as prescribed by emergency procedures. They diagnosed the plant condition and took action necessary to stabilize the unit in a safe condition. The Unit 2 Loop 2 feedwater regulating valve control air tubing was repaired and rerouted to ensure no interferences would result from thermal growth. All other feedwater regulation valves control air tubing was inspected and one other interference issue was resolved.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

B. Corrective Actions to Prevent Recurrence:

Corrective actions include revisions to the conduct of modifications procedures to strengthen the constructability walk down process and thermal growth considerations in piping specifications and installation instructions.

VII. ADDITIONAL INFORMATION

A. Failed Components:

Unit 2 Loop 2 feedwater regulating valve failed closed as a result of a control air tubing break.

B. Previous LERs on Similar Events:

A review of previous reportable events identified a similar event that resulted in a reactor trip that was initiated from a failed feedwater regulating valve control air line.
LER 50-327/95017 addressed the SQN Unit 1 event on December 8, 1995, which resulted from a lack of programmatic controls for maintenance activities that affect vibration through system configuration changes. The January 23, 2007, Unit 2 event is similar in that the thermal movement of the valve was not adequately considered during the planning and implementation of the control air line configuration changes on the Loop 2 feedwater regulating valve. The corrective actions to prevent recurrence from the previous Unit 1 event would not have prevented this air line/fitting failure.

C. Additional Information:

None.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with 10 CFR 50.73(a)(2)(v).

E. Loss of Normal Heat Removal Consideration:

This condition did not result in a loss of normal heat removal.

VIII. COMMITMENTS

None.